ENCLOSURE 2

M210050

Responses to Requests for Additional Information eRAIs 9817, 9826, 9829, and 9831

Licensing Topical Report
NEDC-33922P, Revision 0,
BWRX-300 Containment Evaluation Method

Non-Proprietary Information

IMPORTANT NOTICE

SRP-Review Section: 06.02.01 - Containment Functional Design Application Section:

06.02.01-01 (eRAI 9817)
Date of eRAI Issue: 04/08/2021

Requirement

General Design Criterion 50 – Containment design basis, requires the reactor containment structure, including access openings, penetrations, and the containment heat removal system be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA).

Issue

In order to determine the conservative mass and energy discharge to the containment, a computer code and the associated evaluation model needs to have the capability to model relevant physical phenomenon during a LOCA with a conservative treatment of uncertainties. Standard Review Plan (NUREG-0800) Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," notes that "calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response)."

GEH states in Section 5.2.4 of licensing topical report "BWRX-300 Containment Evaluation Method (NEDC-33922P, Revision 0)," that the only non-condensable gases that may migrate into the isolation condenser system (ICS) tube bundles are the radiolysis products following a design basis LOCA. This is based on the design not experiencing any significant fuel cladding oxidation during a LOCA.

However, based on the BWRX-300 ICS design, [[

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Request

Therefore, the staff is requesting additional information regarding [[
]] the associated modeling uncertainties, and
the subsequent consequences for both large break and small break LOCA limiting cases.

GEH Response to NRC Question 06.02.01-01

It should be noted that the radiolytic gas concentration in the reactor pressure vessel (RPV) is very small even over the long term; therefore, the radiolytic gas concentration at the inlet of the isolation condensers (ICs) is far below the test conditions, such as those used in the PANTHERS tests (see Section 4.2 of Reference R9817-1.).

TRACG has been shown to be in good agreement with the test data as discussed in Section 5.2.4 of NEDC-33922P Revision 0. TRACG accounts for how non-condensable gases (NCGs) degrade the condensation rate as described in Section 6.6.11.2 of Reference R9817-2. The same modeling as approved for the ESBWR PCCS is being used. [[

]] The containment response to large pipe breaks is not affected by any reduction in the IC performance because the RPV isolation valves close long before any buildup of radiolytic gas accumulation could occur. The heat removal rate of the ICs is more than sufficient to maintain the RPV at low pressure after it is isolated, even with severely degraded IC performance. The fuel heat up for small liquid pipe break cases may be affected by degraded IC performance if there is no water injection for three days. In this situation, a reduced IC heat removal rate might result in a higher RPV pressure and faster discharge of RPV inventory. Therefore, the small liquid break case was selected for the sensitivity study.

The comparisons to the small liquid break cases in NEDC-33922P Revision 0 are shown in Figures 9817-1 through 9817-3, below. The red curves are for the expected concentrations of radiolytic oxygen and hydrogen for the conservative case presented in NEDC-33922P Revision 0 whereas the blue curves are for the same conservative case with the concentrations of radiolytic oxygen and hydrogen increased [[   ]]. As shown in Figure 9817-1, there is a small effect on the downcomer level, but the peak cladding temperature still meets the acceptance criteria as shown in Figure 9817-2. The small difference in the liquid level results from [[

]]

As discussed above, the radiolytic gas build up in the IC does not cause a significant degradation in the IC performance because any potential buildup occurs [[

]] Furthermore, the sensitivity calculations demonstrate that the effects of the potential degradation in the IC performance resulting from the presence of radiolytic gases are small even for much higher than reasonably expected radiolytic gas concentrations. The design goal is to limit the radiolytic gas volume fraction in the IC [[   ]]

[[

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Figure 9817-1. RPV Level, Small, Unisolated Liquid Break Sensitivity Study, Conservative Case.

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Figure 9817-2. Peak Cladding Temperature, Small, Unisolated Liquid Break Sensitivity Study, Conservative Case.

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Figure 9817-3. RPV Pressure and Isolation Condenser Efficiency, Small, Unisolated Liquid Break Sensitivity Study, Conservative Case.

References

R9817-1.   NEDC-32725P, "TRACG Qualification for SBWR," Revision 1, August 2002.

R9817-2.   NEDE-32176P, "TRACG Model Description," Revision 4, January 2008.

Proposed Changes to NEDC-33922P Revision 0

None

06.02.01-02 (eRAI 9831)
Date of eRAI Issue: 04/08/2021

BWRX-300 Containment External Surface Thermal Boundary Condition & Shell Modeling

Requirement

Guided by the Standard Review Plan (SRP) Section 6.2.1 and the General Design Criteria (GDCs) 16, 38, and 50 of Appendix A to 10 CFR Part 50 relevant to the containment design basis, the staff is reviewing the applicant's analytical model and assumptions used in the GEH LTR NEDC-33922P, Revision 0, BWRX-300 Containment Evaluation Method. The staff needs to assess the conservatism of the presented model, constitutive/closure relations, model input parameters, and initial/boundary conditions used for the design basis event (DBE) containment response analyses, in order to determine whether the methodology would be acceptably conservative over the applicable range of DBE conditions.

Issue

LTR NEDC-33922P, Section 2.0 states that [[

]]. However, the LTR does not describe the thermal boundary condition assumed for the containment outer surface, except for the PCCS pipes and the containment dome. The LTR is also not clear about whether [[
            ]] is intended to be a standard assumption within the proposed containment evaluation (CE) methodology, or it is just a convenient assumption made for the LTR demonstration analyses. As containment external surface thermal boundary condition and [[
            ]] are key assumptions in meeting several acceptance criteria identified in Section 1.3 of the LTR, the NRC staff would consider them as part of their safety finding about the conservatism.

Request

Therefore, the staff requests a detailed description of the containment outer surface thermal boundary condition as well as [[   ]] assumed as a part of the CE methodology. The applicant is also requested to update the LTR, accordingly.

GEH Response to NRC Question 06.02.01-02

The BWRX-300 containment boundary includes a metal shell. This shell may be free-standing (metal containment type), in loose contact with concrete (reinforced concrete containment vessel type), or in tight contact with concrete (steel concrete composite structure type). Regardless of the eventual containment type, no credit is taken for heat loss from the outer surface of the metal shell to air or concrete (i.e., the containment shell is adiabatic on the outer surface). This is not only an

assumption used in demonstration calculations but is a boundary condition to be used in the application method. This will be clarified in the licensing topical report (LTR).

There are also structures in the containment. The geometry and heat capacity of these structures are currently in development. The demonstration calculations did not credit the energy absorbed in these structures. Neither the composition of the containment shell nor the modeling of internal structures is a limitation on the application method.

Proposed Changes to NEDC-33922P Revision 0

A bullet item will be added to the first set of bullets in Section 6.11 of NEDC-33922P as follows.

•   No credit is taken for heat transfer from the outer surface of the metal containment shell to concrete or surroundings, except for heat transfer from the submerged section of the containment dome to the reactor cavity pool above the dome.

SRP-Review Section: 06.02.01.03 - Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs) Application Section:

06.02.01.03-01 (eRAI 9826)
Date of eRAI Issue: 04/08/2021

Requirement

General Design Criterion 50 – Containment design basis, requires the reactor containment structure, including access openings, penetrations, and the containment heat removal system be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Standard Review Plan (NUREG-0800) Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," identifies 10 CFR Part 50 Appendix K, "ECCS Evaluation Models," as providing the appropriate analysis assumptions and requirements related to sources of energy during the LOCA. SRP 6.2.1.3 further states that calculations of the energy available for release "should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A" and that additional conservatism should be included to maximize the energy release to the containment.

Issue

In this licensing topical report, "BWRX-300 Containment Evaluation Method (NEDC-33922P, Revision 0)," GEH developed a [[

]] considering the specific BWRX-300 dry containment design features.

In contrast to the ESBWR, the BWRX-300 [[

]] Therefore, the staff is requesting additional information from GEH to justify the overall conservatism of the proposed BWRX-300 mass and energy evaluation model.

Request

In particular, additional justification is needed for the following aspects:

1. Isolation Condenser Performance due to accumulating non-condensable radiolytic gas;

2. Core decay heat model using ANS 1979 standard with 2 sigma uncertainty instead of ANS 1971 standard with 1.2 multiplier following the requirement of Appendix K to 10 CFR Part 50;

3. [[

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GEH Response to NRC Question 06.02.01.03-01

The responses to each item are presented below:

1. Accumulation of the non-condensable gases generated from radiolysis is discussed in the response to Question 06.02.01-01 (eRAI 9817). The volume fraction of radiolytic gases in the isolation condensers will be determined after the system requirements for the radiolytic gas removal mechanisms are developed. The method includes a method for calculating radiolytic gas generation and use of a multiplier for the degradation of the isolation condenser heat removal rate as a function of the radiolytic gas volume fraction in the isolation condenser. This method is described in Section 5.2.4 of NEDC-33922P Revision 0. When the method is applied to the eventual isolation condenser design with the radiolytic gas removal mechanisms, the multiplier for the heat removal rate will be set corresponding to the maximum non-condensable gas volume fraction in the isolation condensers.

2. The modeling features in 10 CFR 50 Appendix K provide an acceptable method for demonstrating that the 10 CFR 50.46 acceptance criteria are met for the emergency core cooling systems (ECCS) to maintain fuel integrity. The modeling features in 10 CFR 50.46 do not apply to the BWRX-300 containment analyses because there is no effect of containment performance on the ECCS performance or fuel integrity in the BWRX-300. Section D.2 of Appendix K is not applicable because the containment backpressure is not credited in the BWRX-300 reactor pressure vessel (RPV) mass and energy release calculations and there are no ECCS taking suction from the containment.

    Section 5.3.3 of the NRC final safety evaluation report in Reference R9826-1 acknowledges that the mass and energy release rates used in the BWRX-300 containment analyses will be calculated accounting for all applicable sources of energy required for consideration in 10 CFR 50 Appendix K using the assumptions and correlations similar to those used in Reference R9826-2. The energy sources in 10 CFR 50 Appendix K are listed below along with their applicability to BWRX-300.

    a. Initial stored energy in the fuel: Explicitly accounted for in the TRACG model for each node in the core.

b. Fission heat: A bounding fission power is used as described in Section 5.3.1 of NEDC-33922P Revision 0. Fission power resulting from delayed neutrons is included as part of the decay heat.

c. Decay of actinides: Bounding values are used in the American Nuclear Society (ANS) 5.1-1979 decay heat implementation.

d. Fission product decay: This correlation is obtained from ANS 5.1-1979 + 2 sigma uncertainty, which is the same as in Section 3.2.4 of Reference R9826-2 as stated above.

e. Metal-water reaction rate: Metal-water reaction is precluded in BWRX-300 design basis accidents by requiring that the fuel cladding temperature remain below normal operating temperature.

f. Reactor internals heat transfer: This is explicitly calculated as part of the TRACG method.

g. Pressurized water reactor primary-to-secondary heat transfer: Not applicable to BWRX-300.

Section 5.2.5 of NEDC-33922P Revision 0 uses the decay heat and the decay heat uncertainty used in the previously approved Reference R9826-2 (Reference 7.10 of NEDC-33922P Revision 0). The bounding cases in Reference R9826-2 also use nominal + 2 sigma uncertainty values by compounding the conservatisms rather than combining them using a statistical method, which is the same as in NEDC-33922P Revision 0. The applicability of Reference R9826-2 to BWRX-300 is discussed in Sections 5.1 and 5.2 of NEDC-33922P Revision 0.

3. The base cases use nominal feedwater temperature, but the conservative cases use either the nominal or reduced feedwater temperature as shown in Table 5-2 of NEDC-33922P Revision 0. The uncertainty in the feedwater temperature is very small (see Section 6.2.2 of Reference R9826-3). However, plants may operate with a reduced feedwater temperature as an equipment out-of-service condition. The flow rate from a liquid pipe break increases with reduced feedwater temperature. At BWRX-300 operating conditions, this increase in flow rate has a larger effect on the total energy release to the containment than the reduction in the enthalpy. Therefore, both the mass release and the energy release from a feedwater pipe break is higher if the feedwater temperature is decreased. As shown in Table 5-2 in the NEDC-33922P Revision 0, reduced feedwater temperature is used for the liquid pipe break conservative cases.

Reduced feedwater temperatures do not have an adverse effect on the steam pipe break mass and energy release. The liquid at the top of the downcomer above the feedwater spargers flashes when the pressure decreases because this fluid is saturated regardless of the feedwater temperature. With reduced feedwater temperature, flashing and level swell become less pronounced. [[

]] As a result, the normal feedwater temperature case is the more limiting case although the difference from the reduced feedwater temperature case is insignificant for steam pipe breaks.

4. As shown in Section 8 of Reference R9826-3, the effect of the steam dome pressure is small on the overall RPV response to a pipe break. The BWRX-300 technical specifications do not yet exist, and therefore the allowable values have not yet been established. A dome pressure of 20 psi higher than the normal pressure is a bounding value for the pressure controller uncertainties and is a typical value for the difference between the normal operating pressure and the technical specification allowable value for conventional plants. If a limit is established in the BWRX-300 technical specifications for the allowable dome pressure similar to the conventional plants, the application methodology will use the dome pressure in the limiting conditions for operation in Modes 1 and 2 in the BWRX-300 technical specifications. Otherwise, the upper limit of the pressure controller uncertainty band will be used for the initial pressure.

5. As shown in Section 8.1 of Reference R9826-3 and also as stated in its safety evaluation, the initial water level does not have a significant effect on the LOCA analyses. Large break cases, which are rapidly isolated, use normal water level. However, the initial RPV water inventory may be important in the small unisolated break cases in BWRX-300 because there is a coping period of three days without injection. In order to bound the potential variations in the initial RPV inventory, the low end of the normal operating range is used for small breaks as shown in Table 5-2 of NEDC-33922P Revision 0.

6. In the conservative cases, a two-sigma uncertainty bias is applied to the critical flow rate. The bias applied to the critical flow rate was shown to adequately bound the scatter in the data in Section 3.4.1 of Reference R9826-2. The critical flow model and the bias applied to it are the same as those in the previously approved Reference R9826-2 for ESBWR LOCA and containment analyses.

7. It was not necessary to construct a new phenomena identification and ranking table (PIRT) and identify the biases and uncertainties for BWRX-300 RPV phenomena, because the ESBWR PIRT and uncertainties are directly applicable to BWRX-300 as discussed in Sections 5.1 and 5.2 of NEDC-33922P Revision 0. The correlations used for the phenomena in the BWRX-300 RPV are the same as those in the ESBWR that have been previously reviewed and approved in Reference R9826-2. In the conservative case, the biases for these parameters are all applied in the conservative direction simultaneously rather than combining them using a statistical method.

References

R9826-1.   NEDC-33911P-A, "BWRX-300 Containment Performance," Revision 2, April 2021.

R9826-2.   NEDC-33083P-A, "TRACG Application for ESBWR," Revision 1, September 2010.

R9826-3.   NEDE-33005P-A, "TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6," Revision 2, May 2018.

Proposed Changes to NEDC-33922P Revision 0

None

SRP-Review Section: 06.02.04 - Containment Isolation System Application Section: 1.3

06.02.04-01 (eRAI 9829)
Date of eRAI Issue: 04/08/2021

Requirement

General Design Criterion 50 – Containment design basis, requires the reactor containment structure, including access openings, penetrations, and the containment heat removal system be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Issue

LTR NEDC-33922P, Section 1.3, "Acceptance Criteria," states in one of the criteria that "containment remains isolated for 72 hours during a design basis event or accident." The staff found that this statement might not be valid in some accidents such as isolation condenser system LOCA outside containment. It is understood that the subjects of containment isolation and pipe failures outside containment are not in the scope of this LTR.

Request

The applicant is requested to clarify the statement by specifying the applicability of the statement within the LTR and applicable conditions.

GEH Response to NRC Question 06.02.04-01

The purpose of the last acceptance criterion listed in Section 1.3, which is the subject of this eRAI, is to specify that the other acceptance criteria should be met without venting the containment for at least 72 hours. This is not a requirement for the containment operation but a requirement for the acceptable demonstration of containment performance.

Proposed Changes to NEDC-33922P Revision 0

To clarify the purpose of this statement, the last bulleted item in Section 1.3 will be made a paragraph and modified as follows:

"Containment venting or leakage shall not be credited for at least 72 hours in demonstrating that the above acceptance criteria are met during a design basis event or accident."